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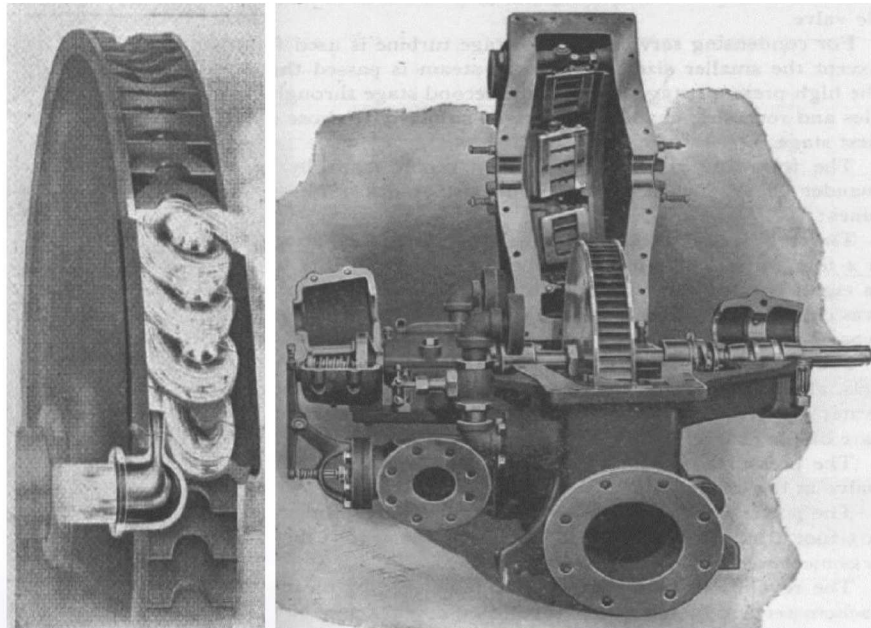
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Terry Turbopump Analytical Modeling Efforts in Fiscal Year 2016 – Progress Report

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Abstract

This document details the Fiscal Year 2016 modeling efforts to define the true operating limitations (margins) of the Terry turbopump systems used in the nuclear industry for Milestone 3 (full-scale component experiments) and Milestone 4 (Terry turbopump basic science experiments) experiments. The overall multinational-sponsored program creates the technical basis to: (1) reduce and defer additional utility costs, (2) simplify plant operations, and (3) provide a better understanding of the true margin which could reduce overall risk of operations.

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ACRONYMS

BDBE	Beyond Design Basis Event
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owner's Group
CST	Condensate Storage Tank
DOE	U.S. Department of Energy
DOE-NE	U.S. Department of Energy's Office of Nuclear Energy
EOP	Emergency Operating Procedure
EPG	Emergency Procedure Guidance
EPRI	Electric Power Research Institute
ExOB	Expanded Operating Band
HPCI	High Pressure Coolant Injection
IAE	Institute of Applied Energy
INL	Idaho National Laboratory
INPO	Institute of Nuclear Power Operations
LWRS	Light Water Reactor Sustainability
METI	Government of Japan's Ministry of Economy, Trade, and Industry
NEI	Nuclear Energy Institute
NEUP	Nuclear Energy University Programs
NRC	U.S. Nuclear Regulatory Commission
NSIAC	Nuclear Strategic Issues Advisory Committee
Turbo-TAG	Nuclear Grade Terry Turbopump Advisory Group
PIM	Pooled Inventory Management
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
PWROG	Pressurized Water Reactor Owner's Group
QA	Quality Assurance
RCIC	Reactor Core Isolation Cooling
RCP	Reactor Coolant Pump
RHR	Residual Heat Removal
RSMC	Risk Management Subcommittee
RST	Reactor Safety Technologies
RPV	Reactor Pressure Vessel
SAG	Severe Accident Guidance
SG	Steam Generator
SNL	Sandia National Laboratories
TAMU	Texas A&M University
TDAFW	Turbine Driven Auxiliary Feedwater
TTUG	Terry Turbine User Group

1. Introduction

This section provides the motivation for Sandia National Laboratories' (SNL) efforts to assist the world-wide commercial nuclear power community in characterizing the behavior of the reactor core isolation cooling (RCIC) system for boiling water reactors (BWRs) and turbine driven auxiliary feedwater (TDAFW) system for pressurized water reactors (PWRs) under beyond design basis operations. Also, this section provides background information, and a discussion on the analytical model used for this work, MELCOR.

1.1 Purpose and Motivation

The Fukushima accident demonstrated both the challenges associated with severe accident management, and the importance of understanding the behavior of critical equipment under beyond design basis conditions. The purpose of this project is to improve reactor safety for emergency and severe accident management by understanding real-world performance of critical components (i.e., experimental testing and analytical modeling will allow for RCIC/TDAFW to be more accurately characterized under beyond design basis (e.g., station blackout with an extended loss of AC power conditions). The current use of conservative assumptions regarding equipment functioning as found in probabilistic risk assessment (PRA) applications limits the anticipated prevention and mitigation options considered for emergency operation procedures (EOPs) and severe accident guidelines (SAGs). This work is part of an overall project (***Terry Turbine Expanded Operating Band Summary of Program Plan – Revision E***) that would experimentally test and analytically verify the RCIC/TDAFW steam-driven Terry turbopump performance under beyond design basis event (BDBE) conditions. This project would be jointly funded through support from the U.S. Department of Energy's Office of Nuclear Energy (DOE-NE), U.S. nuclear industry, and international stakeholders.

The overall goal of the project is to understand the real-world behavior of Terry turbopump operation under BDBE conditions in order to advance our predictive fidelity and applicability in emergency and severe accident prevention and mitigation. Accurate characterization of the RCIC/TDAFW system could have fleet-wide impacts in how EOPs and SAGs will be implemented (e.g., knowing a RCIC pump will last longer than an hour or two after DC power is lost will allow operators to consider other options for plant recovery or accident mitigation). Further, investigation of turbopump performance may also provide insights into means for improving severe accident performance (e.g., accident tolerant fuels).

The purpose of this research is to further develop a dynamic and mechanistic system-level model of the RCIC/TDAFW turbine/pump system capable of predicting the system performance under BDB conditions that include two-phase water ingestion into the Terry turbine at various potential reactor operating pressures, and to characterize its ability (or not) to maintain adequate water injection with sufficient pump head under degraded operating conditions. This work is a continuation of SNL efforts in 2015 [1]. The model discussed in Section 2.1 will also demonstrate the self-regulating mode of operation as was observed in the Fukushima Daiichi Unit 2 accident, where RCIC ran uncontrolled and successfully maintained reactor water inventory for nearly three

days. Section describes 2.2 aspects of two-phase flow anticipated for a provisional MELCOR model analysis of TDAFW operation in beyond design basis conditions.

This work is provides additional insights for developing a thermodynamically-based analytical model of the steam-driven RCIC/TDAFW system operation with mechanistic accounting of liquid water carryover and pump performance degradation, to be used in codes like MELCOR or MAAP. These insights will provide the basis for experimental design to operate a Terry turbopump under extended uncontrolled operating conditions. The scaled and full-scale Terry turbopump experiments will support an improved understanding of plant risk, improve plant operations, and provide the technical basis for improving the reliability of an essential plant system as shown in the three main categories below¹:

1. **Regulatory/Risk:** Test data can reduce plant operational risk and improve regulatory compliance
 - Improved incident response timing and prediction of RCIC/TDAFW performance to determine staffing needed to implement beyond design basis mitigation activities
 - Improved response to regulatory changes associated with post Fukushima Lessons Learned
 - A better prediction of the core damage frequency reduction associated with implementation of beyond design basis mitigation activities
2. **System Improvement:** Improve system reliability; operation of an essential system needed to mitigate/prevent risk dominate accidents
 - Identifies RCIC enhancements and changes in maintenance practices to meet Fukushima Lessons Learned
 - Provides performance data on refurbished hardware (including I&C)
 - Provides for system performance conditions for station blackout (SBO)-like conditions to allow for proper quantification of needed system margins
3. **Plant Operations:** Improves operations during an BDBE to mitigate the accident under a wide range of plant conditions
 - Identifies optimal approaches to operate RCIC/TDAFW during a long-term SBO and loss of heat sink
 - Provides data to support identification of RCIC/TDAFW performance conditions could complicate or challenge FLEX implementation
 - Identification of proper handoff conditions from RCIC/TDAFW to FLEX

¹ Letter from BWROG to DOE-NE Federal Programs Manager Richard A. Reister, BWROG-14066, dated November 21, 2014.

1.2 Background

Prior to the accidents at Fukushima Daiichi, modeling of the performance of key critical components such as the RCIC/TDAFW steam-driven Terry turbopump and safety relief valves (SRVs) are based mostly on design basis conditions. Their performance under severe accident conditions is poorly known and largely based on conservative assumptions used in PRA applications. For example, common PRA practice holds that battery power (DC) is required for RCIC operation to control the BWR reactor pressure vessel (RPV) water level, and that loss of DC power results in RCIC flooding of the steam lines. The flooding of the steam lines is assumed to lead to a subsequent failure of the RCIC system due to two-phase water ingestion into the turbine-side of the pump. This assumption for accident analysis implies that RCIC operation should terminate on battery depletion which can range from between 4 hours and 12 hours [2]. In contrast, real-world observation from Fukushima Unit 2 shows that RCIC function was affected but not terminated by uncontrolled steam line flooding, and in fact provided coolant injection for almost three days [3]. Similar issues and uncertainties exist for PWRs as well with the use of the TDAFW system to feed steam generators (i.e., the same steam-driven Terry turbopump is used for RCIC and AFW systems).

Use of conservative assumptions regarding equipment functioning as found in PRA applications may limit the anticipated mitigation options considered for emergency operations and severe accident management procedures. Improvements to reactor safety can be realized for severe accident management if real-world performance of critical components such as the RCIC steam-driven turbine pump can be more faithfully characterized. Improved understanding of this critical component can be realized through a combination of advanced modeling methods such as embodied in the DOE/Industry sponsored CASL project and through scaled and large scale testing.

The purpose of this research is to develop a dynamic and mechanistic system-level model of the Terry turbopump for the RCIC/TDAFW system which is capable of predicting the system performance under beyond design basis conditions that include two-phase water ingestion into the Terry turbine at various potential reactor operating pressures, and to characterize its ability (or not) to maintain adequate water injection with sufficient pump head under degraded operating conditions. This model will also demonstrate the self-regulating mode of operation as was observed in the Fukushima Daiichi Unit 2 accident, where RCIC ran uncontrolled and successfully maintained reactor water inventory for nearly three days.

1.3 Analytic Tools

Several analytical tools are being applied to investigate RCIC/TDAFW behavior for severe accidents. The tools include reactor system modeling codes such as MELCOR and RELAP, in addition to computational fluid dynamic (CFD) codes such as FLUENT and SolidWorks Flow. The primary goal is a mechanistic, system-level model that permits fast execution of long transient simulations (i.e. several hours to days for severe accidents). This will enable simulation capabilities for Fukushima forensic analyses, the development of technically-defensible SAG/FLEX strategies, and design analysis of potential upcoming Terry turbopump experiments. The intent of using several codes, both system-level and CFD, is to inform and enhance the system-

level modeling efforts using focused CFD analyses of key components, particularly where lumped-parameter methods and simple hand calculations have limited capability. An example is CFD analysis of the steam nozzles that drive the Terry turbine. The computer code being applied in the RCIC/TDAFW modeling is briefly described in the following subsection.

1.3.1 MELCOR

MELCOR is a fully integrated, engineering-level computer code that models the progression of severe accidents in light-water reactor nuclear power plants [4]. MELCOR is being developed at SNL for the U.S. Nuclear Regulatory Commission (NRC) as a second-generation plant risk assessment tool, and the successor to the Source Term Code package. A broad spectrum of severe accident phenomena in both BWRs and PWRs is treated in MELCOR in a unified framework. These include thermal-hydraulic response in the reactor coolant system, reactor cavity, containment, and confinement buildings; core heat-up, degradation, and relocation; core-concrete attack; hydrogen production, transport, and combustion; fission product release and transport behavior. MELCOR applications include estimation of severe accident source terms, and their sensitivities and uncertainties in a variety of applications. Design basis accidents in advanced plant designs (e.g., the Westinghouse AP-1000 design and the GE Hitachi Nuclear Energy ESBWR design) have been analyzed with MELCOR.

Current applications of MELCOR include the NRC sponsored State-of-the-Art Reactor Consequence Analyses (SOARCA) [2], and the U.S. Department of Energy (DOE) sponsored Fukushima Daiichi accident analyses [3].

2. Terry Turbopump System Response

Recent SNL RCIC system modeling has centered on including a mechanistic representation of a RCIC turbine/pump in a comprehensive model of the Fukushima Daiichi Unit 2 reactor and containment system [1]. Until this recent effort, mechanistic RCIC modeling had been confined to an otherwise coarse model of Fukushima Daiichi Unit 2 laden with manipulations of boundary conditions that substituted for detailed representations of the reactor, drywell and wetwell. The coarse model served initial concept-based modeling of RCIC well but lacked needed realism as RCIC modeling matured. In Section 2.1, a presentation follows of the results obtained in modeling the accident at Fukushima Daiichi Unit 2 with MELCOR utilizing detailed representations of the reactor, drywell and wetwell and employing a mechanistic representation of a RCIC turbine/pump. New insights have resulted from the modeling and new testing needs have become apparent. The testing needs are identified in context and expanded upon in Section 3. Section 2.2 provides an assessment of the TDAFW system for unregulated Terry turbopump conditions and is a system-level analysis similar to that done for the initial RCIC assessment [1].

2.1 RCIC Modeling

In the course of accomplishing the subject modeling, a key action on the part of the Fukushima Unit 2 operators was recognized that had not been previously recognized. The action was an early throttling of RCIC water delivery to the RPV accomplished by partially circulating flow back to the condensate storage tank (CST). The action was taken approximately 45 minutes into the accident (i.e., 45 minutes after the earthquake) to stop RCIC from cycling between on and off as it had been doing in response to RPV level becoming too low or too high, respectively. The throttling reduced RCIC flow to the RPV by roughly half as evidenced in the Fukushima Unit 2 data shown in Figure 1. It is plausible that the partial diversion of RCIC flow to the CST was not stopped when RCIC suction was switched from the CST to the wetwell by the operators 10.75 hours into the accident. Consequently, wetwell inventory may have been continually pumped to the CST from the time of the switchover to the time RCIC stopped operating. It should be noted that whether the partial routing of RCIC flow to the CST was stopped at switchover or not, it would have been immaterial with respect to the ability of RCIC to self-regulate and cool the reactor. As evidenced in modeling results presented below, the RCIC turbine/pump would have simply moved to differing states supplying virtually the same flow to the reactor in either case.

An ongoing question concerning the response of the Fukushima Unit 2 RCIC system has been, “Why didn’t the system over speed and trip when its speed controller failed upon losing electrical power approximately 1 hour into the accident.” An understood consequence of a loss of power to a RCIC controller is a movement of the RCIC turbine steam admission (governor) valve to the fully open position. Previous SNL modeling has suggested that the increased steam flow resulting from a fully open governor valve would easily drive a RCIC system so that it would over speed and mechanically trip. Clearly that did not happen at Fukushima Unit 2, implying that something was lacking in the modeling. SNL currently wonders if what is lacking is a good understanding of how the reversing chambers installed in the turbine casing benefit turbine operation. Specifically, there is a suspicion that the benefit of the chambers is a strong function of turbine speed with greatest benefit at zero speed; substantial benefit at rated speed and negligible benefit

at speeds greater than rated. *Testing is needed to prove/disprove this hypothesis.* In the modeling presented, a complete loss of benefit from the reversing chambers was assumed at rated speed (4,500 rpm) plus 50 rpm. This kept RCIC system from rising much above the typical mechanical trip speed of 125% of rated speed (5,625 rpm).

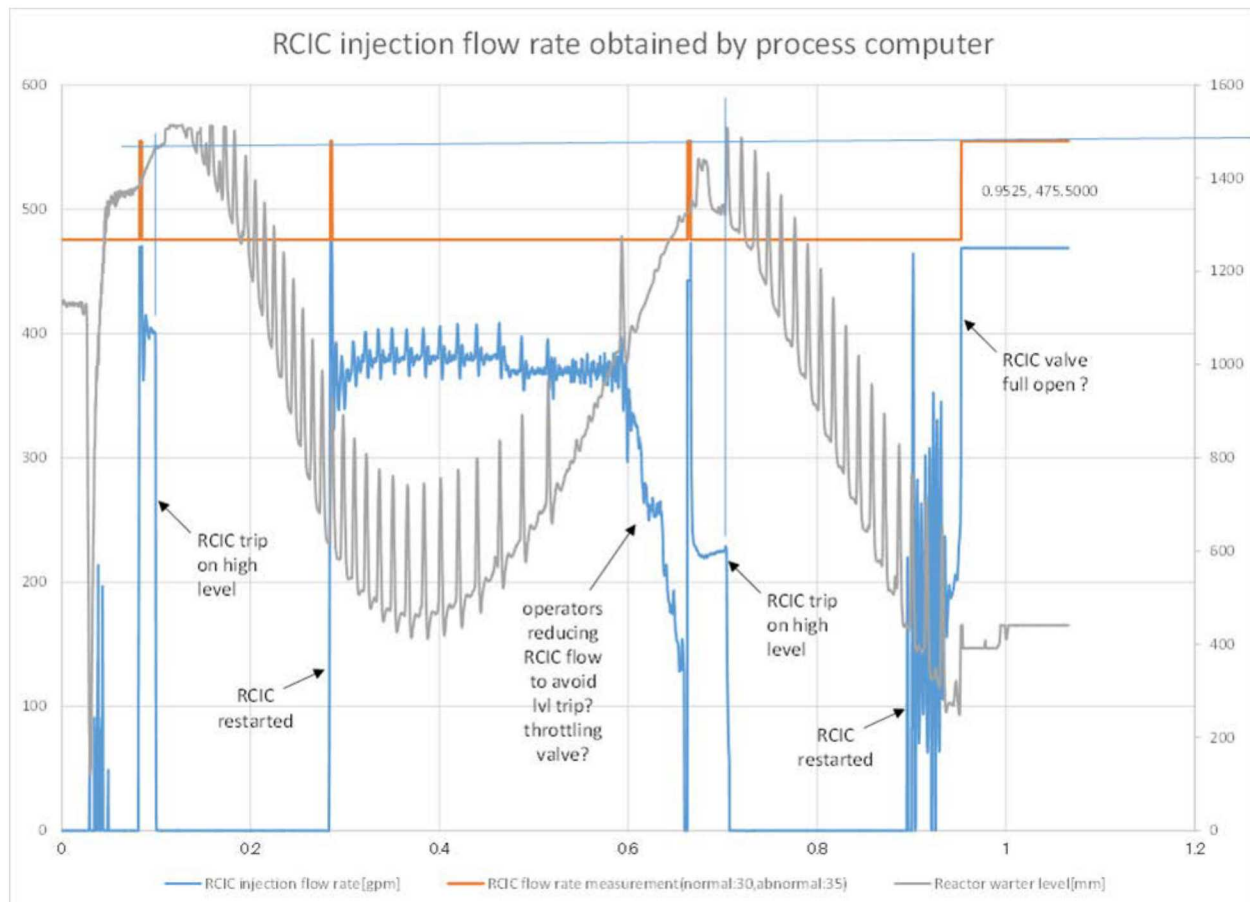


Figure 1 RCIC Flow to the RPV and RPV Level Recorded Early in Fukushima Unit 2 Accident

Figure 2 and Figure 3 show the speed response of the RCIC system in the MELCOR modeling. Figure 4 shows the associated pressure history of the RPV. Following the speed controller's loss of electrical power and resulting fully-open condition of the turbine governor valve, RCIC speed, flow and power increased dramatically. For this analysis, only the assumed loss of benefit for the reversing chambers at speeds above rated speed kept the RCIC system from accelerating to speeds that could trip the turbine overspeed. RCIC water delivery to the RPV was maintained, even with the circulation of water back to the CST described above. This increased the water level to where it exceeded what was required to remove decay heat. Consequently, the RPV flooded to the main steam line (MSL) nozzles and spilled over into the MSLs. Water entered the piping that would normally supply saturated steam to the RCIC turbine and the turbine took on water. Water pooled in the turbine steam ring covering some nozzles (lower nozzles) such that they could flow only water. Uncovered (higher) nozzles continued to flow steam, but the loss of steam flow through

the covered nozzles substantially reduced the driving force available to produce turbine power. This pooling phenomenon was considered in the MELCOR modeling, where nozzles situated lower in the turbine steam ring became submerged. *This needs to be validated through testing.* Figure 5 and Figure 6 show RCIC flow. Figure 7, Figure 8, and Figure 9 show RCIC pump head, RCIC turbine/pump power and RPV level, respectively. The pooling of water in the RCIC turbine steam ring is shown in Figure 10 and the effective number of five total RCIC turbine nozzles flowing steam/water are shown in Figure 11. Figure 12 shows the pressure response along the RCIC turbine steam supply and exhaust piping. Figure 2 through Figure 12 exemplify the self-regulating RCIC system operation evidenced to have occurred at Fukushima Unit 2.

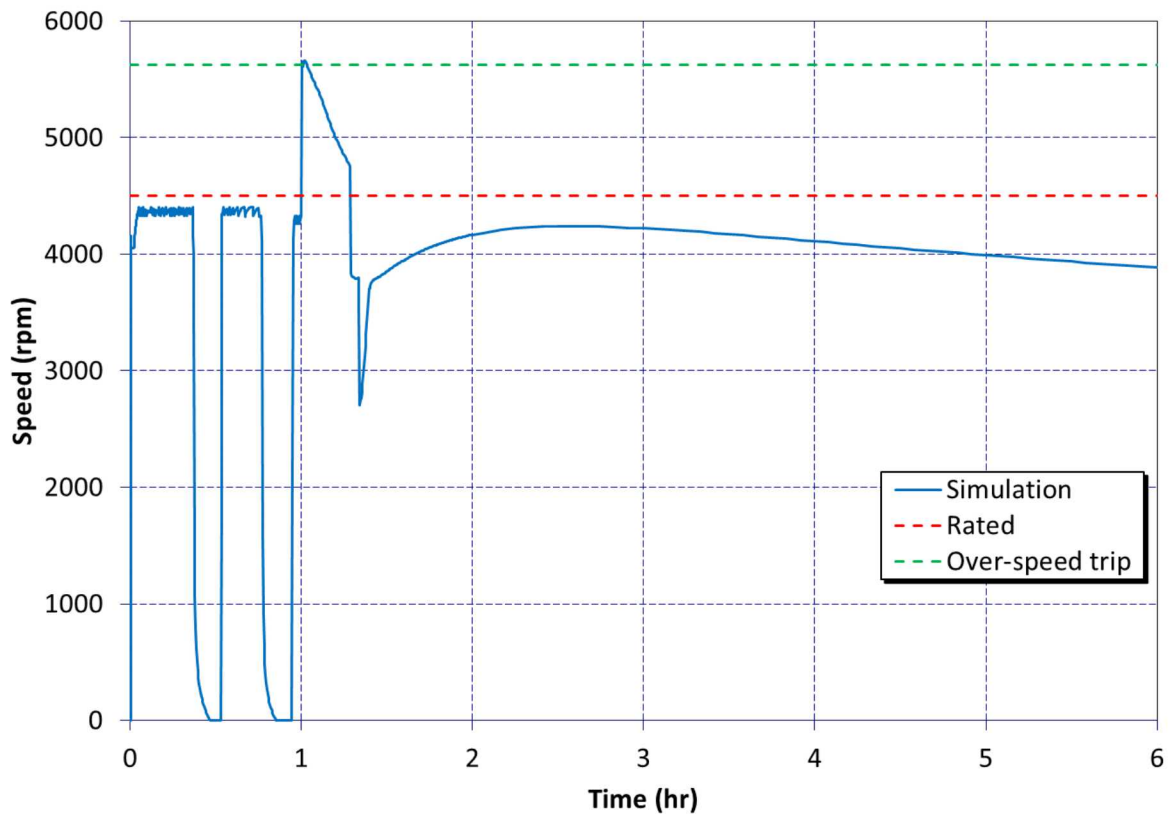


Figure 2 RCIC Speed to 6 hours

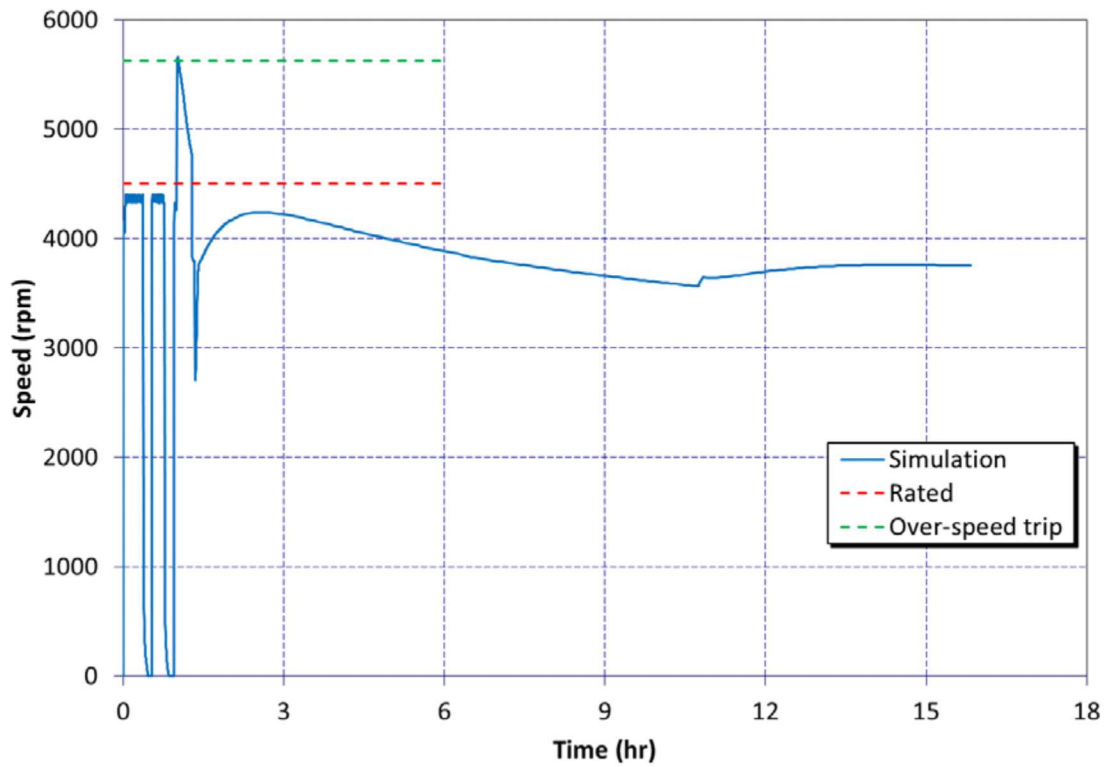


Figure 3 RCIC Speed to 18 hours

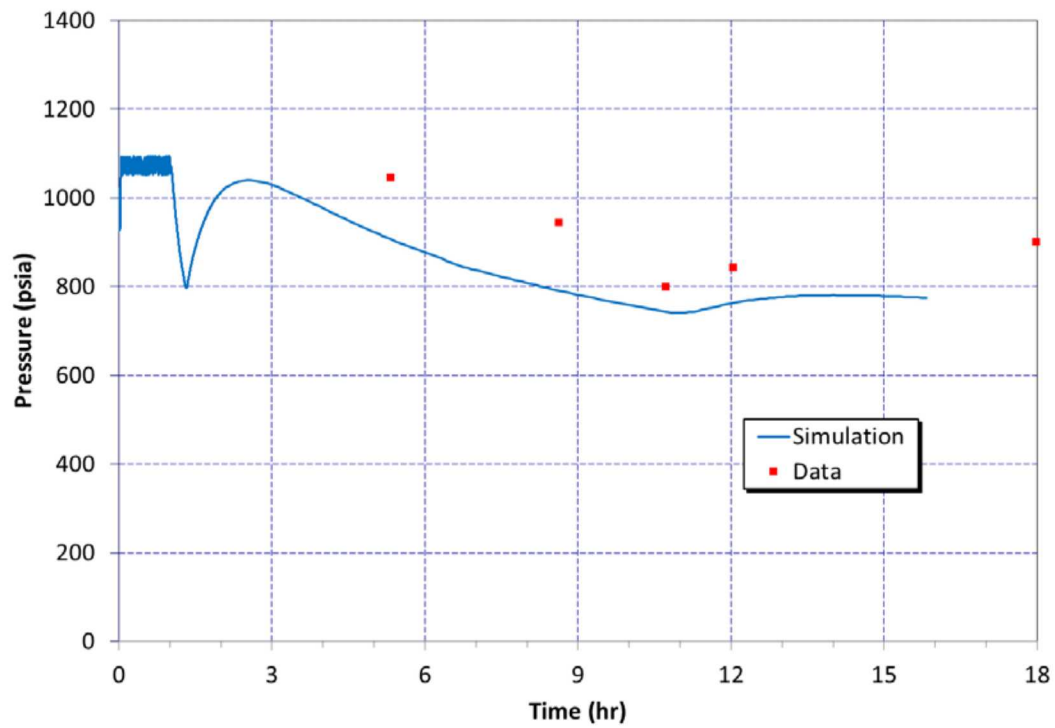


Figure 4 RPV Pressure

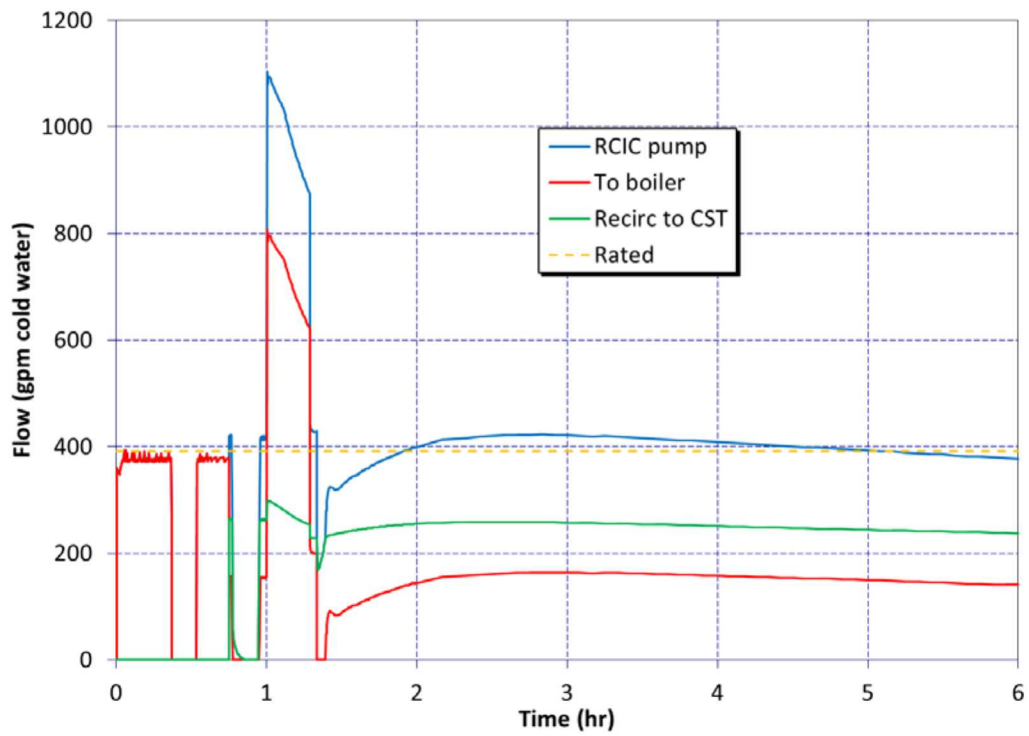


Figure 5 RCIC Flow to 6 hours

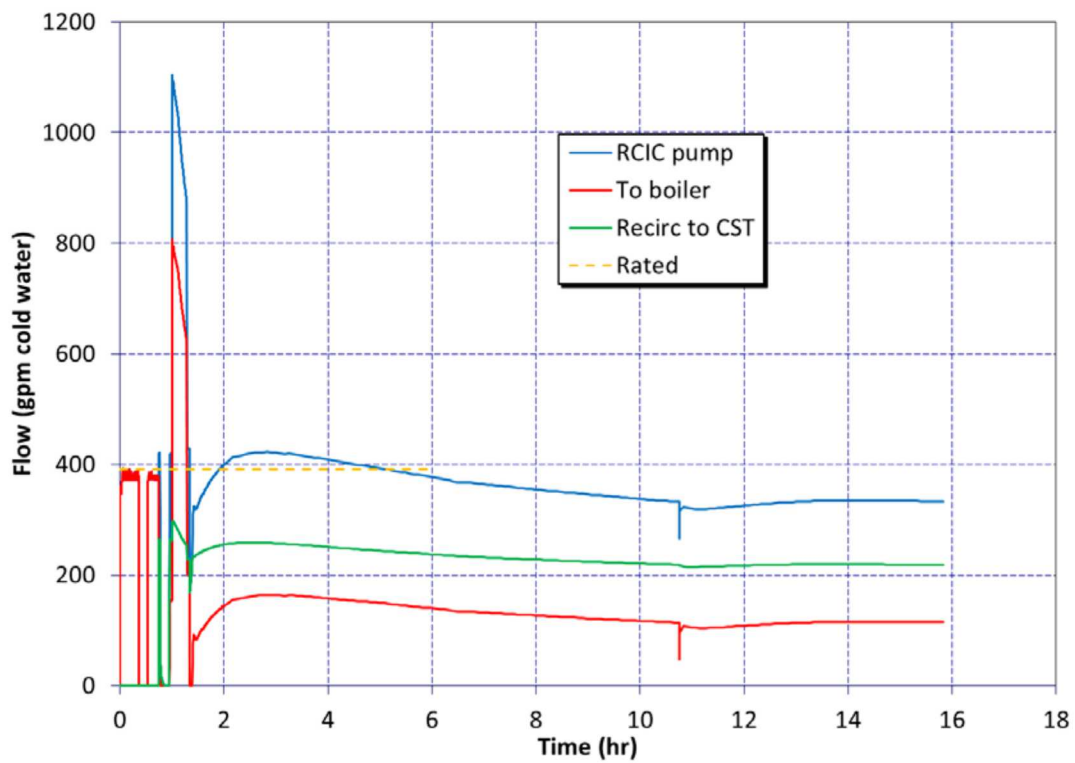


Figure 6 RCIC Flow to 18 hours

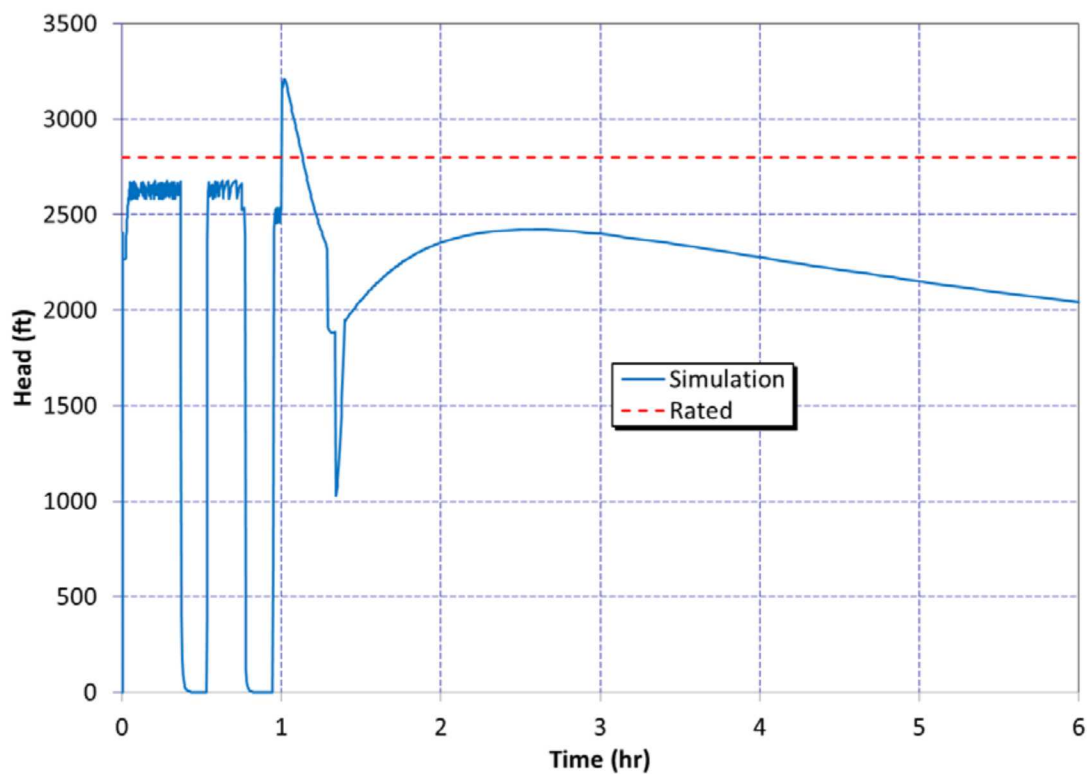


Figure 7 RCIC Pump Head

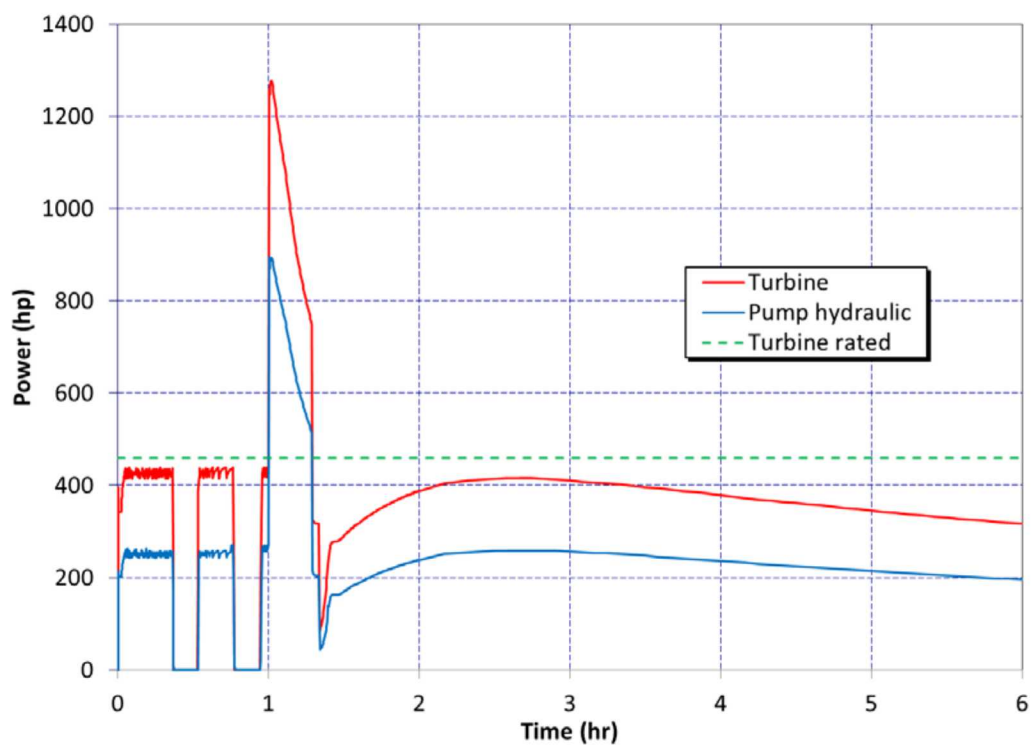


Figure 8 RCIC Turbine Power and Pump Hydraulic Power

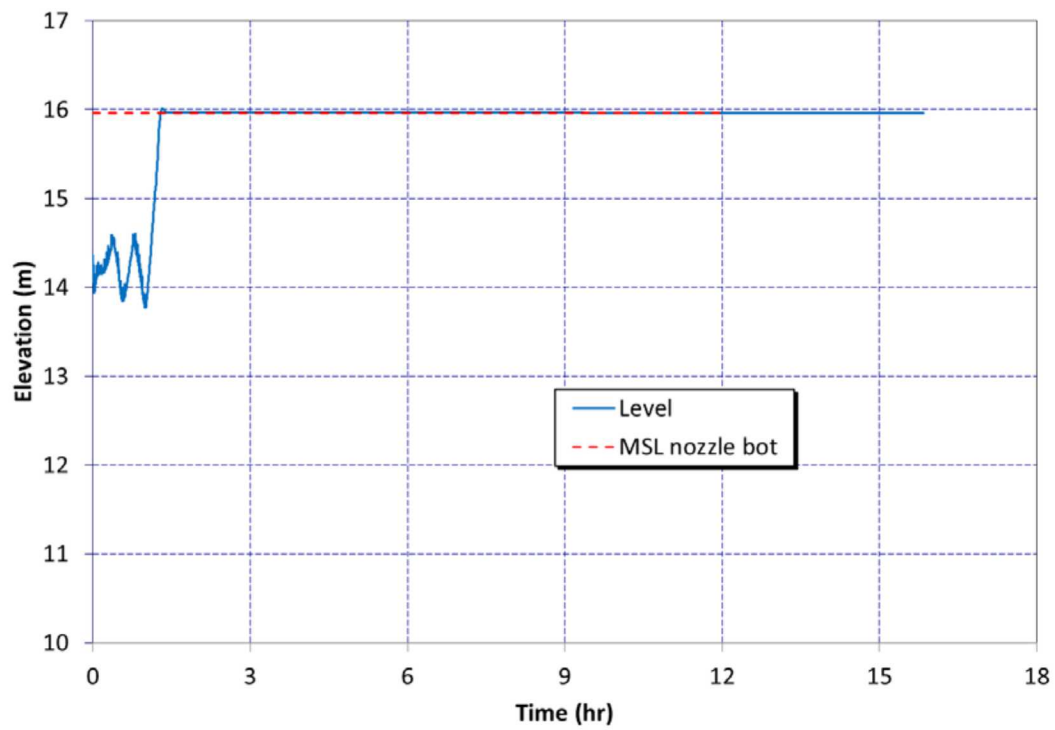


Figure 9 RPV Level

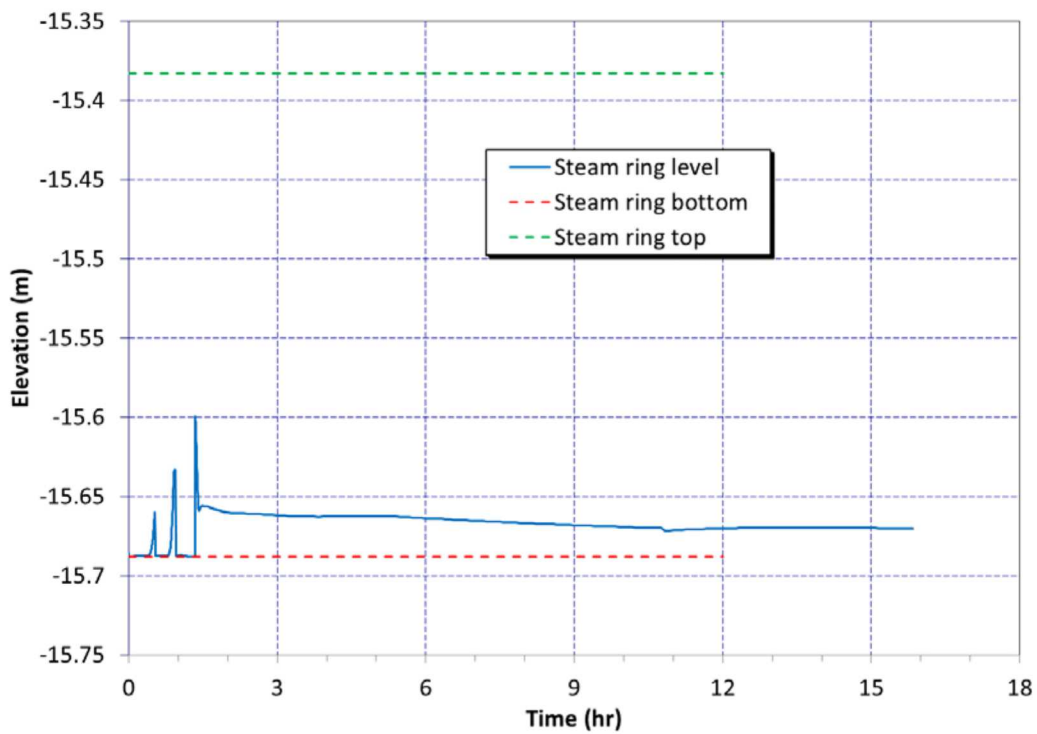


Figure 10 Water Pooling in RCIC Turbine Steam Ring

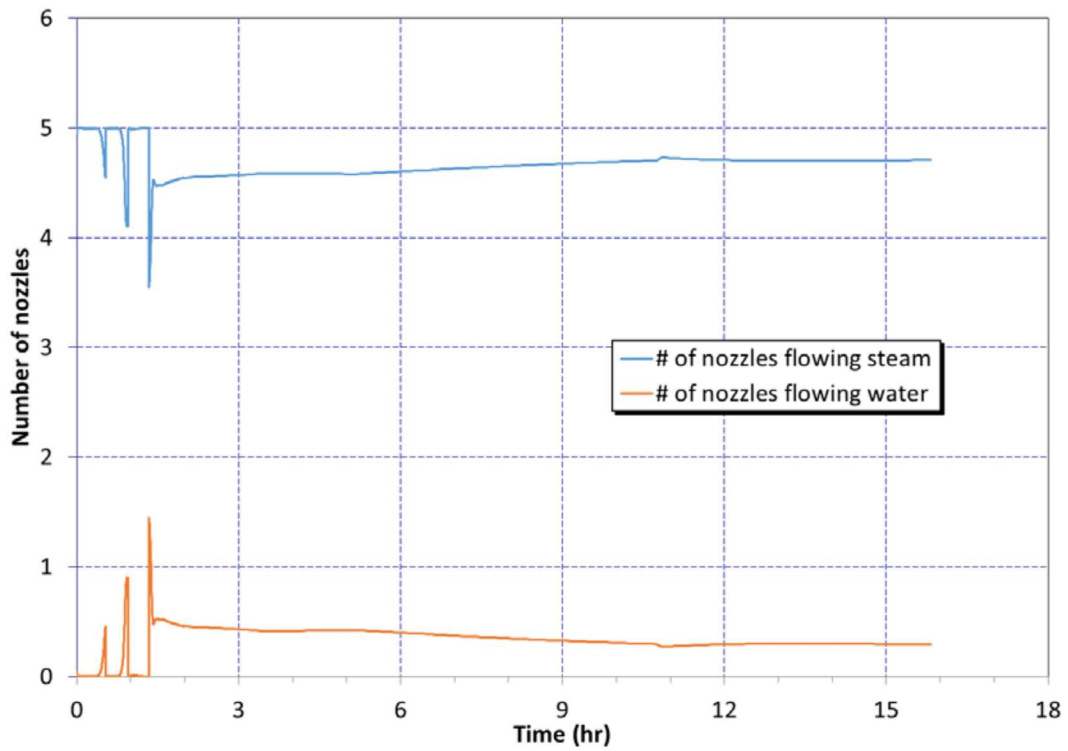


Figure 11 Number of RCIC Turbine Nozzles Flowing Steam/Water (Five Total)

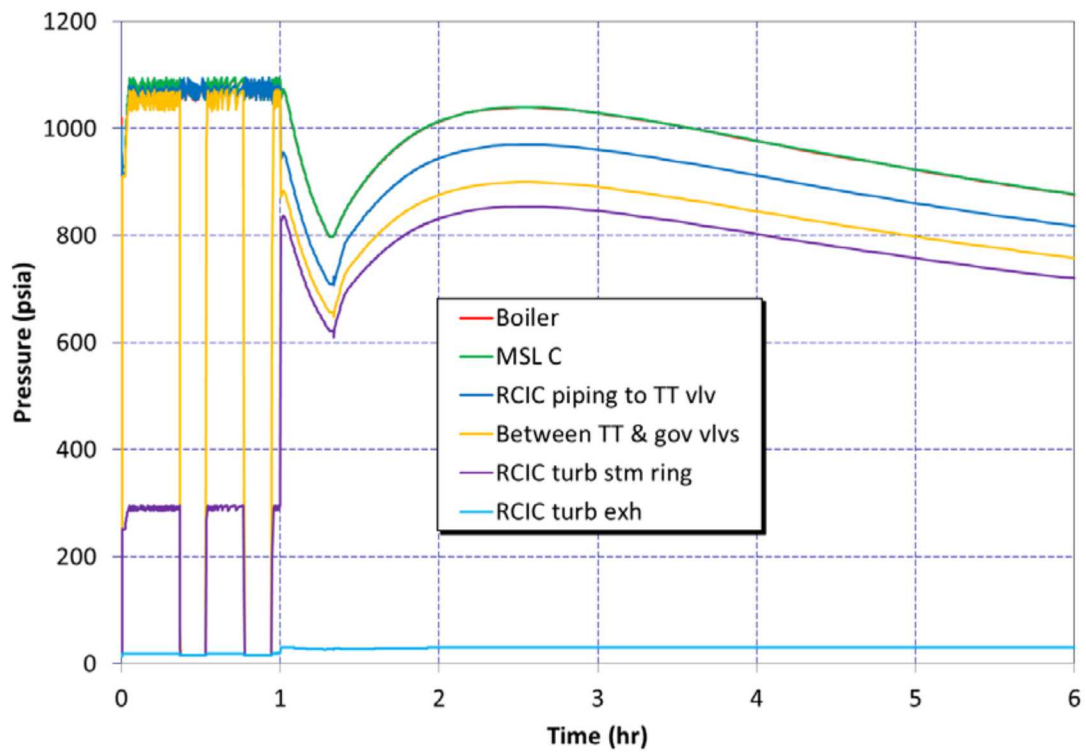


Figure 12 Pressure along RCIC Turbine Steam Supply and Exhaust Piping

A further consequence of a RCIC turbine taking on water is the deleterious effect of water accumulation in the turbine casing (the enclosure in which the turbine wheel resides/spins). Water in the casing would hinder the normal supersonic expansion of steam jetting from the nozzles which is important to turbine power production. Additionally, the turbine wheel would drag as it spun through water accumulated in the casing. Water flooded the control volume representing the turbine casing and exhaust piping in the MELCOR calculation. The vertical rise of approximately 23 feet in the Fukushima Unit 2 turbine exhaust piping from the outlet of the turbine to the high point of the piping facilitated the flooding. The deleterious effect of accumulated water in a RCIC turbine casing is accounted for in the current RCIC modeling by simply including a degradation multiplier of 0.5 on the turbine torque developed by steam flow. This value of the multiplier gives a fair correspondence in reactor pressure history between the Fukushima Unit 2 modeling and Fukushima Unit 2 data while allowing RCIC to deliver water to the RPV continuously. It should be noted that the value of this multiplier is not critical in the modeling with respect to whether RCIC self-regulates to cool the reactor. For instance, a value of 0.75 simply lowers the pressure at which RCIC self-regulates, while a value of 0.25 shows RCIC flow to the reactor interrupts and reestablishes repeatedly as the RCIC steam supply line floods and clears cyclically. The latter situation may be what developed at Fukushima Unit 2. *Testing is needed to determine if this is true – testing to better understand the deleterious effect of accumulated water in a RCIC turbine casing.*

Figure 13 shows the velocity of steam entering and leaving a bucket on the turbine wheel and the velocity of a water jet approaching a bucket relative to the tangential velocity of turbine wheel. Notice in this figure is that water velocity is lower than wheel tangential velocity meaning that water jetting from the nozzles could not be driving the wheel. The buckets of the wheel, in fact, would be slapping the jetting water causing a deleterious effect on turbine power. Figure 14 reflects this effect showing the torque produced by jetting water as negatively valued.

The inflections seen in the RPV pressure trace of Figure 4 and RCIC flow trace of Figure 6 when RCIC suction switched from the CST to the wetwell at 10.75 hours (as evidenced in RPV pressure data) are in response to the elevated temperature of the water in the wetwell relative to the temperature of the water in the CST. Less heat per unit flow is needed to heat the warmer wetwell water to saturation. Thus in the RCIC self-regulating situation, more flow is needed to cool the reactor and more steam pressure is needed to drive the larger flow.

The modeling leading to the pressure, flow and pooling results of Figure 4, Figure 6, and Figure 10 assumed the partial routing of RCIC flow to the CST was not stopped when operators switched RCIC suction from the CST to the wetwell. Figure 15, Figure 16, and Figure 17 compare these results to results obtained when the partial routing is instead stopped at switchover such that all RCIC flow is delivered to the reactor. Notice that RCIC pump flow changes considerably, but flow delivery to the RPV does not. RPV pressure is the same. More water pools in the RCIC turbine steam ring further covering nozzles. The RCIC pump/turbine simply moves to a different state to cool the reactor with close to the same flow.

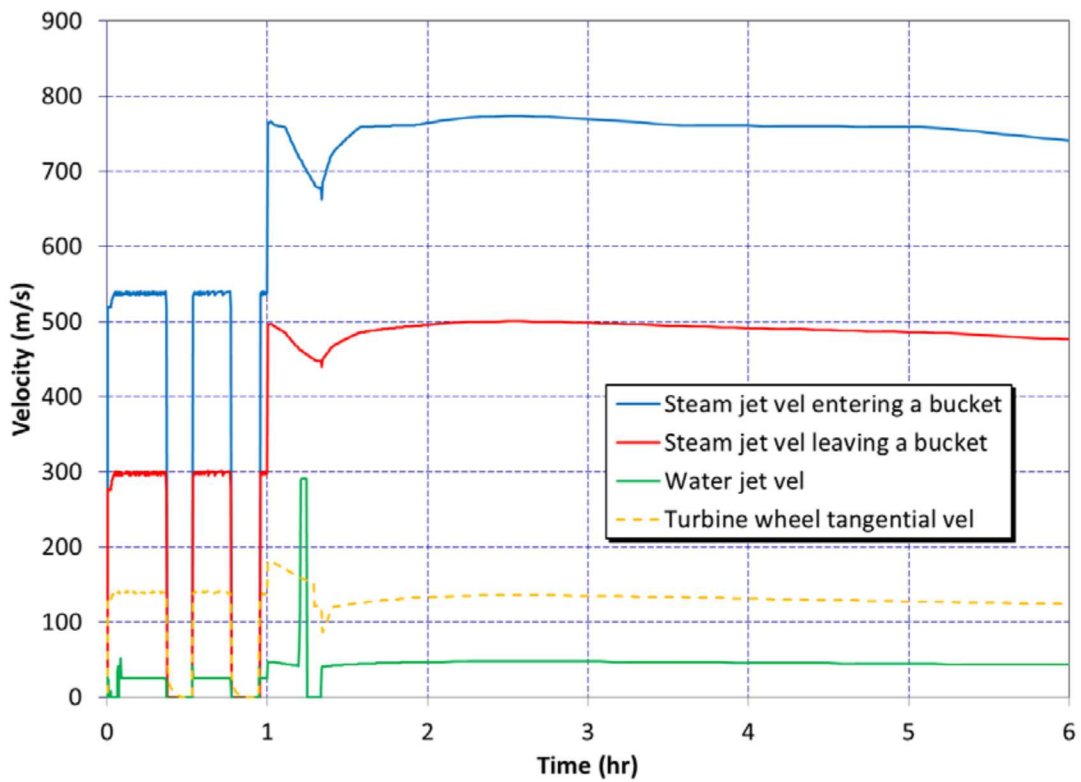


Figure 13 Steam/Water Jet Velocity Relative to Turbine Wheel Tangential Velocity

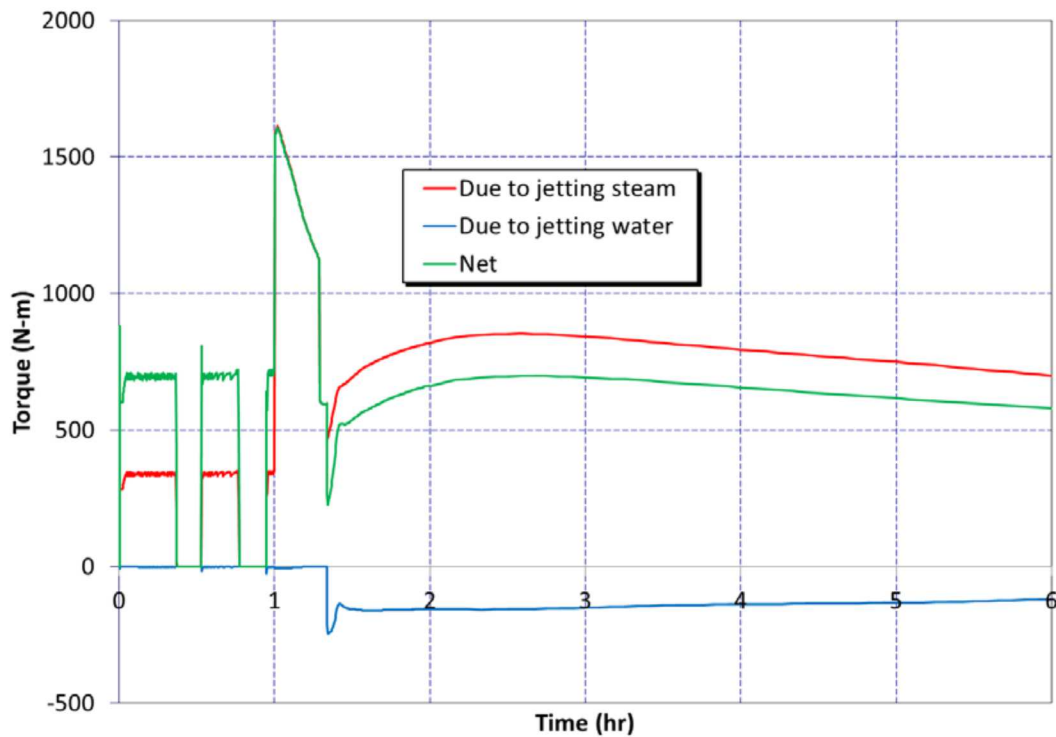


Figure 14 Turbine Torque Produced by Jetting Steam/Water

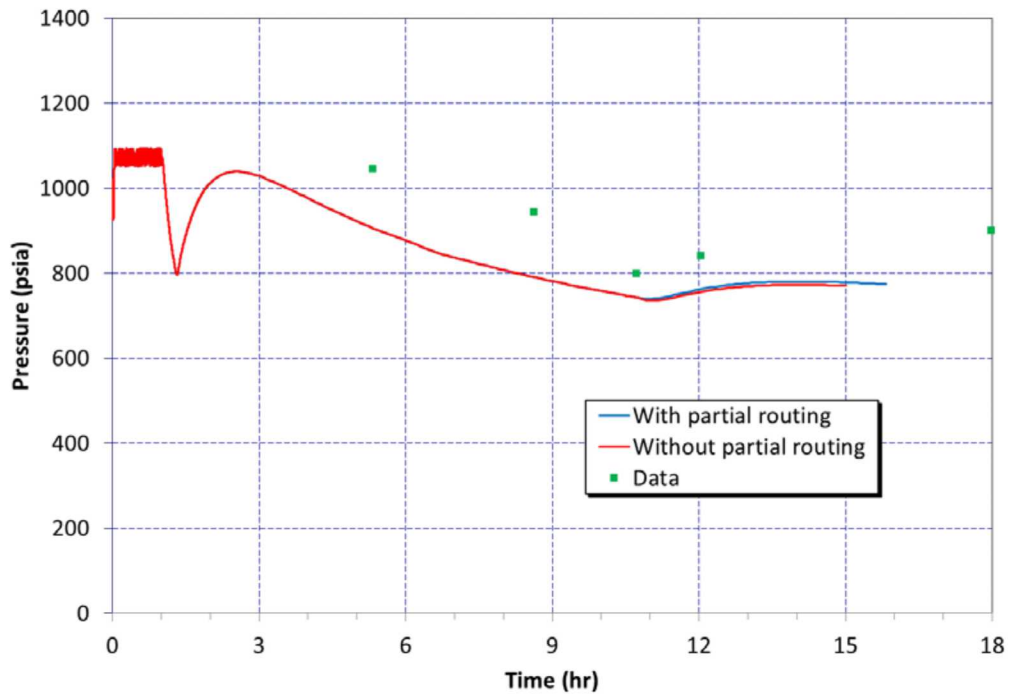


Figure 15 RPV Pressure with/without Partial Routing of RCIC Flow to CST after Switchover

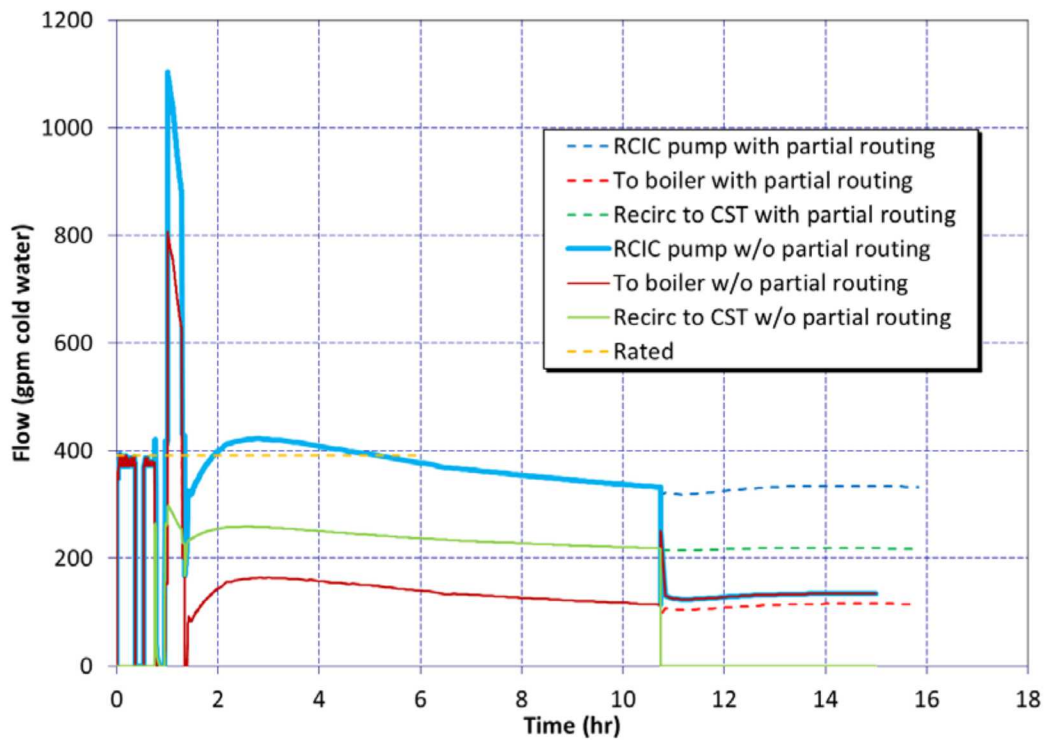


Figure 16 RCIC Flow with/without Partial Routing of RCIC Flow to CST after Switchover

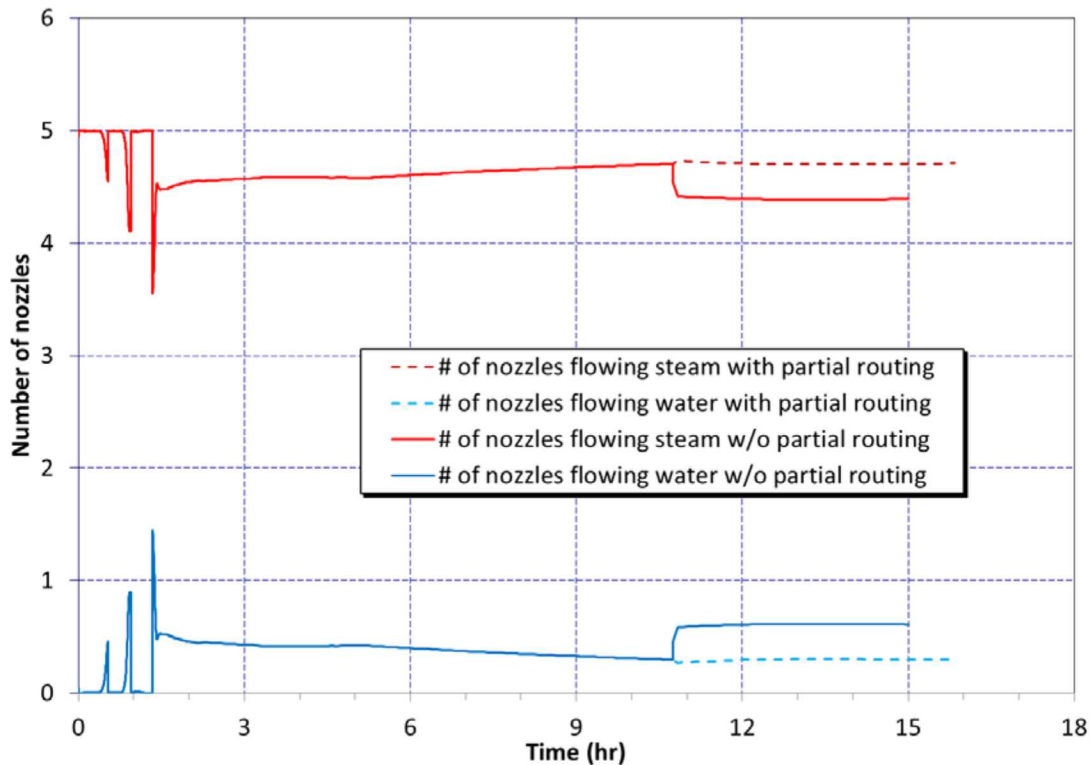


Figure 17 RCIC Turbine Steam Ring Pooling with/without Partial Routing of RCIC Flow to CST after Switchover (Five Total)

When RCIC eventually failed approximately 66 hours into the accident, steam flow to the turbine may well have continued. The only occurrence that could have stopped the flow would have been a closing of the turbine trip-throttle valve resulting from a mechanical overspeed trip. Given that an overspeed trip did not happen earlier when the RCIC turbine speed controller lost electrical power and the overspeed threat was high, it seems unlikely that an overspeed trip was the reason for ultimate failure of the RCIC system.

2.2 TDAFW Modeling

The PWR MELCOR model used to analyze the TDAFW system response is based on the Surry SOARCA analysis [5] with the addition of the Terry turbopump model developed for the initial RCIC system response for the SNL effort in 2015 [1] with a 32-hour simulation time. This analysis assumes a long-term SBO with only one-hour of DC battery power and a reactor coolant pump (RCP) seal leakage of 21 gpm at normal operating primary pressure. Figure 18 provides the primary and secondary pressure response for 0 to 5 hours, and also shows when the accumulators begin injection once primary pressure reaches ~600 psi. For similar reasons discussed in Section 2.1, the overspeed trip of the Terry turbine is neglected when DC power is lost as shown in Figure 19; the Terry turbine does not approach the overspeed trip setpoint at any other time of the simulation.

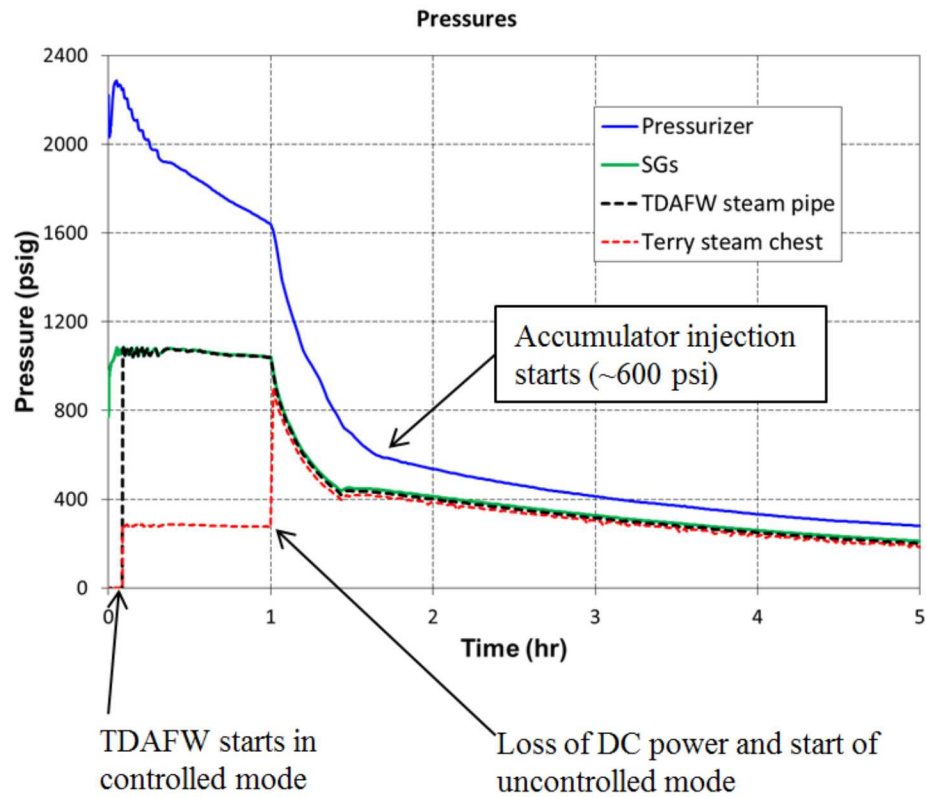


Figure 18 Primary and Secondary Pressure Response to 5 hours

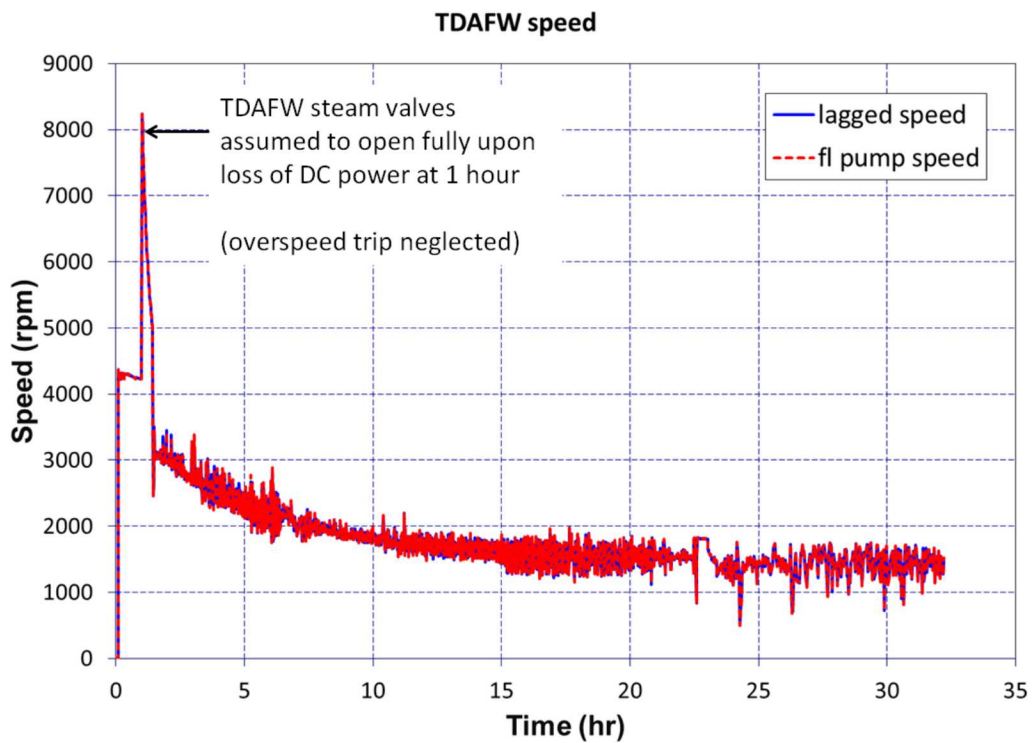


Figure 19 TDAFW Turbine Speed

The system-level model TDAFW solution utilizing the homologous pump features is intended to be the solution carried forward in future TDAFW modeling work. The solution is expected to add needed realism for Terry turbopump self-regulating simulations and will inform the design of full-scale testing configurations. For this 32-hour simulation, the following insights were determined from the observed modeling data:

- The primary did not decouple from the secondary due to the assumed RCP seal leakage for the 32-hour simulation. This indicates that unregulated TDAFW may provide a significant period of time for adequate core cooling. Cooldown of the primary system by the TDAFW system reduces primary leakage (e.g., RCP seal leakage) prolonging the effective coupling between the primary and secondary.
- Due to RCP seal leakage and exhaustion of accumulator injection, loss of primary water inventory will ultimately result in steam generator (SG) tube voiding as shown by Figure 20. If a sufficient number of SG tubes are voided, there will be a loss of adequate core cooling since the primary will experience a loss of natural circulation.
- The accumulators will provide a sufficient amount of primary water injection for hours, but will ultimately become exhausted, ~15 hours as seen in Figure 21 and Figure 22. Once the accumulators are exhausted, there is no primary system makeup water and ultimately the primary will become decoupled from the secondary due to the loss of natural circulations (i.e., steam voiding in the SG tubes). However, this was not observed for this simulation.

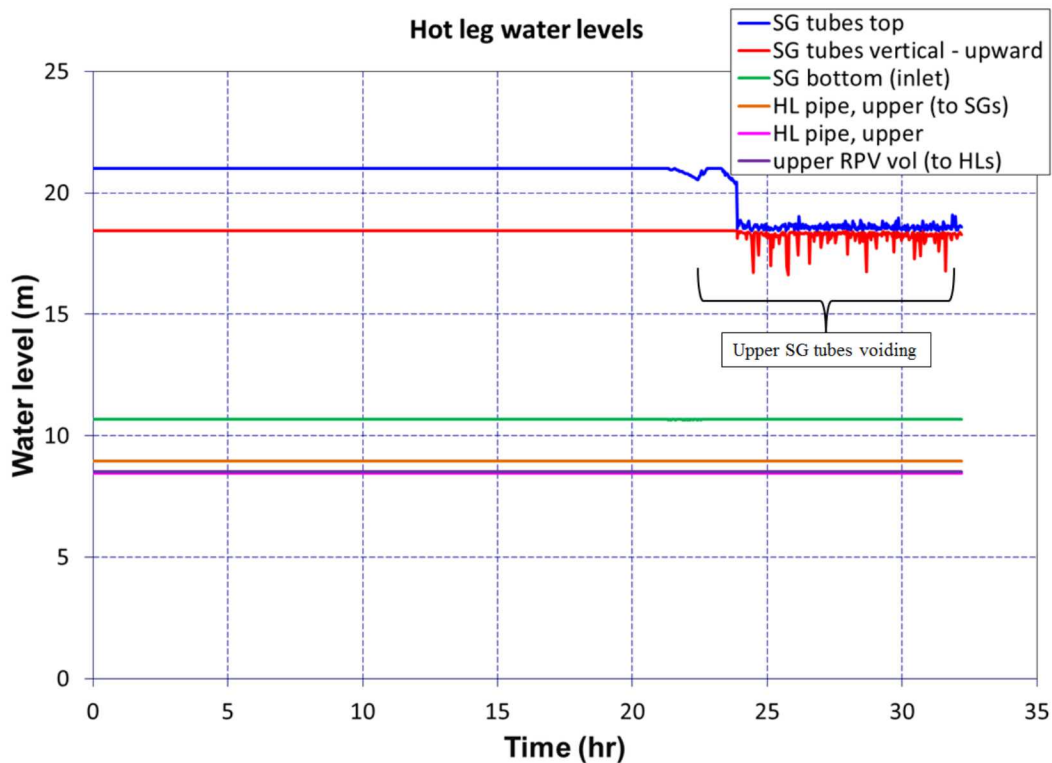


Figure 20 Primary-Side Steam Generator and Hot Leg Water Level

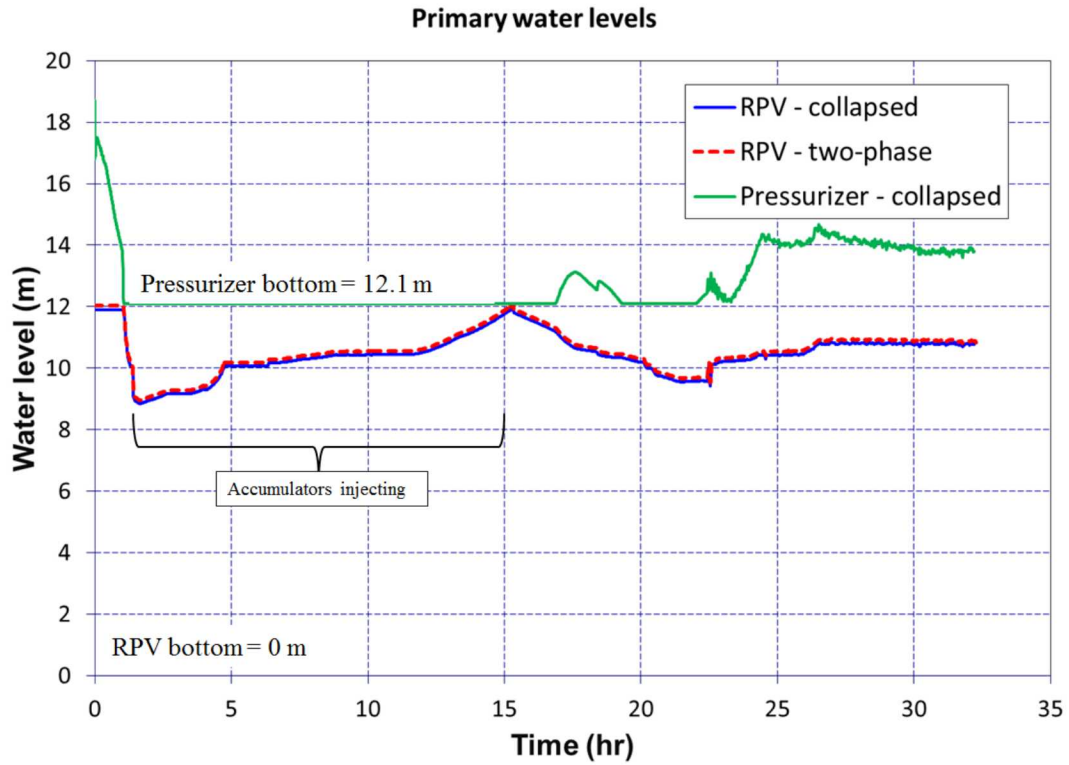


Figure 21 RPV and Pressurizer Water Level

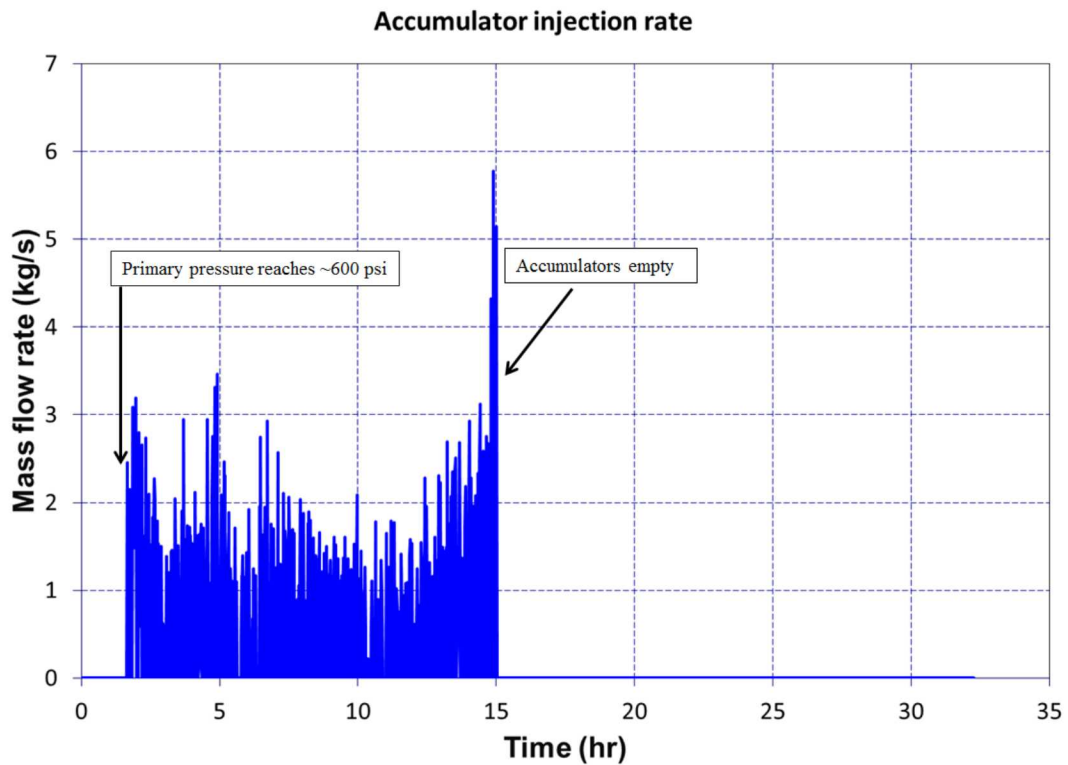


Figure 22 Accumulator Injection Rate

It is apparent that without experimental testing data, there is no data for the validation of this system-level TDAFW model. This further makes the case for some level of experimental testing to assure within validated boundary conditions TDAFW models will properly predict and inform the nuclear industry in updating emergency operating procedures, establishing a technical basis for operational changes that can prevent progression to core damage (i.e., reduce core damage frequency), and simplifying plant operations by increasing the time available for implementation of FLEX.

3. Suggested Terry Turbopump Testing to Inform Modeling

The described modeling of the Fukushima Unit 2 accident suggests certain experimental testing on Terry turbines that could greatly further understanding of the operation of these machines in nominal and off nominal conditions. The following describes in brief what the testing might best involve.

To investigate the variability in the benefit of reversing chambers with turbine speed:

- With a Terry turbine loaded by a dynamometer, exercise the turbine throughout its operating speed range and record the power developed.
- Remove the reversing chambers from the turbine and again exercise the turbine throughout its operating range.
- Compare the horsepower versus speed (rpm) curves of the two tests to determine the benefit of the reversing chambers overall and the dependency of the benefit with speed. Look in particular for any marked drop in benefit at speeds above rated.

To investigate the detrimental influence of water accumulation in the turbine steam ring:

- With water introduced to the steam or air driving the turbine and with the exhaust piping serving the turbine configured such that the casing of the turbine can readily drain, investigate how power drops off with increasing water introduction.
- Investigate whether water pools in the steam ring covering some of the nozzles. Crucial in this testing is that the turbine casing readily drains. Recall, the casing of the turbine is the enclosure in which the turbine wheel resides/spins.

To investigate the detrimental influence of water accumulation in the turbine casing:

- With water introduced to the steam or air driving the turbine and with the exhaust piping serving the turbine configured such that the casing of the turbine cannot readily drain, investigate how power drops off as water accumulates in the casing.

4. References

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